New Reactors Programme

GDA close-out for the AP1000 reactor

GDA Issue GI-AP1000-FS-06 – Validation of the IRWST Cooling Function for the PRHR

Assessment Report: ONR-NR-AR-16-026
Revision 0
March 2017

Published 03/17

EXECUTIVE SUMMARY

Westinghouse Electric Company LLC (Westinghouse) is the reactor design company for the **AP1000®** reactor. Westinghouse completed Generic Design Assessment (GDA) Step 4 in 2011 and paused the regulatory process. It achieved an Interim Design Acceptance Confirmation (IDAC) which had 51 GDA Issues attached to it. These issues require resolution prior to the award of a Design Acceptance Confirmation (DAC) and before any nuclear safety-related construction can begin on site. Westinghouse re-entered GDA in 2014 to close the 51 GDA Issues.

This report is the Office for Nuclear Regulation's (ONR's) assessment of the Westinghouse **AP1000** reactor design in the area of fault studies. Specifically, this report addresses GDA Issue GI-AP1000-FS-06: Validation of the IRWST Cooling Function for the PRHR.

The key safety innovation identified by Westinghouse for the **AP1000** reactor is the provision of a Passive Core Cooling System (PXS) to provide core cooling following design basis accidents. A notable aspect of the PXS is the Passive Residual Heat Removal (PRHR) heat exchanger. The PRHR heat exchanger is located in the In-containment Refuelling Water Storage Tank (IRWST) at an elevation above the reactor core. Following a design basis reactor fault in which the primary circuit remains intact, the PRHR heat exchanger plays a vital role in decay heat removal, transferring heat from the Reactor Coolant System (RCS) into the IRWST. This transfer of heat causes the water in the IRWST to heat up, eventually becoming saturated, and initiates steaming from the tank. The steam released from the IRWST condenses on the inner surface of the containment vessel, giving up heat originating in the RCS, and (by design) forms a thin fluid film of water which runs down the inner containment wall surface. Provisions are made to collect and channel condensate to the IRWST, replenishing the steam losses and allowing the passive heat removal process to continue.

This GDA Issue arose in GDA Step 4 because Westinghouse provided no detailed justification in its safety case documentation for its assumptions on how much of the condensate would be returned to the IRWST. Some of the steam will condense and get trapped on other structures within the containment. Alternatively, it could drain to sumps in the basement of the containment, bypassing the IRWST. Over a period of time (that time being dependent on how big the condensate losses are), the PRHR heat exchanger could uncover and cease to be effective.

It was recognised by Westinghouse that the underlying technical concern was applicable to all **AP1000** plants (ie it is an issue that is not unique to the UK) and during the period of time it had paused its GDA interactions, it was actively engaged in addressing this issue to the satisfaction of US and Chinese customers and regulators. Westinghouse's approach to closing this GDA Issue has therefore involved a UK-specific report and an update to the UK **AP1000** Pre-Construction Safety Report (PCSR) but these are supported by generic reports, calculations and test results which have also been shared with overseas regulators.

Westinghouse has needed to:

- undertake two programmes of physical testing to better understand the relevant phenomena;
- make several significant design changes to the **AP1000** design to increase the amount of condensate returning to the IRWST;
- develop a new analysis methodology to model the behaviour of the **AP1000** plant in the applicable fault scenarios;
- identify a limiting fault and demonstrate that all relevant safety acceptance criteria are met for a 72-hour transient; and
- modify its definition of what represents a safe, stable shutdown state for the **AP1000** reactor and place a time limit on how long that state can be maintained with the PXS.

I am satisfied that the extensive scope of work delivered by Westinghouse is adequate and addresses the requirement of the GDA Issue. In coming to this judgement, I have been

informed by an ONR-commissioned report written by a Technical Support Contractor who has looked in detail at Westinghouse's submissions. I have also made extensive use of a publicly available evaluation report written by the United States Nuclear Regulatory Commission (US NRC) on the same underlying technical concern. By not repeating the assessment work of others, I have been able to focus my regulatory attention on the aspects of the submissions which are unique to the UK safety case.

In summary, I am satisfied that GDA Issue GI-AP1000-FS-06 can be closed.

LIST OF ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurisation System
ALARP	As Low As Reasonably Practicable
ATWS	Anticipated Transients Without Scram
CMT	Core Makeup Tank
DAC	Design Acceptance Confirmation
DCP	Design Change Proposal
DNBR	Departure from Nucleate Boiling Ratio
EDCD	European Design Control Document
EPRI	Electric Power Research Institute
GDA	Generic Design Assessment
IAEA	International Atomic Energy Agency
IDAC	Interim Design Acceptance Confirmation
IRWST	In-containment Refuelling Water Storage Tank
LOCA	Loss of Coolant Accident
MDEP	Multinational Design Evaluation Programme
NNSA	National Nuclear Safety Administration (of China)
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation
PCCWST	Passive Containment Cooling Water Storage Tank
PCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PIRT	Phenomena Identification and Ranking Table
PRHR	Passive Residual Heat Removal
PXS	Passive Core Cooling System
RCS	Reactor Coolant System
SAP	Safety Assessment Principle
SSC	Structures, Systems and Components
TSC	Technical Support Contractor
US NRC	United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators Association

TABLE OF CONTENTS

1 INTRODUCTION

1.1 Background

1. Westinghouse Electric Company LLC (Westinghouse) is the reactor design company for the **AP1000®** reactor. Westinghouse completed Generic Design Assessment (GDA) Step 4 in 2011 and paused the regulatory process. It achieved an Interim Design Acceptance Confirmation (IDAC) which had 51 GDA Issues attached to it. These issues require resolution prior to the award of a Design Acceptance Confirmation (DAC) and before any nuclear safety-related construction can begin on site. Westinghouse re-entered GDA in 2014 to close the 51 GDA Issues.

2. This report is the Office for Nuclear Regulation's (ONR's) assessment of the Westinghouse **AP1000** reactor design in the area of fault studies. Specifically, this report addresses GDA Issue GI-AP1000-FS-06: Validation of the IRWST Cooling Function for the PRHR.

3. The related GDA Step 4 report (Ref. 1) is published on our website (**www.onr.org.uk/new-reactors/ap1000/reports.htm**), and this provides the assessment underpinning the GDA Issue. Further information on the GDA process in general is also available on our website (**www.onr.org.uk/new-reactors/index.htm**).

1.2 Overview of GI-AP1000-FS-06

4. The key safety innovation identified by Westinghouse for the **AP1000** reactor is the provision of a Passive Core Cooling System (PXS) to provide core cooling following design basis accidents. The PXS is designed for core residual heat removal, safety injection, and depressurisation without the use of active equipment such as pumps and AC power sources.

5. A notable aspect of the PXS is the Passive Residual Heat Removal (PRHR) heat exchanger. The PRHR heat exchanger is located in the In-containment Refuelling Water Storage Tank (IRWST) at an elevation above the reactor core. The inlet to the heat exchanger is connected to one of the two hot legs of the primary circuit while the outlet is connected to the outlet plenum on one of the two steam generators (the steam generator on the loop with the hot leg connection). Following a design basis reactor fault where the primary circuit remains intact and there is not a Loss of Coolant Accident (LOCA), the PRHR heat exchanger plays a vital role in decay heat removal, transferring heat from the Reactor Coolant System (RCS) into the IRWST.[1] This transfer of heat causes the water in the IRWST to heat up, eventually becoming saturated, and resulting in steaming from the tank.

6. Eventually, the heat in the steam needs to be transferred to an ultimate heat sink. In the case of the **AP1000**, this is the outside atmosphere. This is achieved by another passive safety system supporting the PXS, the Passive Containment Cooling System (PCS). This feature is designed to provide heat removal from the containment shell to the environment via natural circulation of air and evaporative cooling of water flowing from the Passive Containment Cooling Water Storage Tank (PCCWST) under gravity. The steam released from the IRWST condenses on the inner surface of the containment vessel, giving up heat originating in the RCS, and (by design) forms a thin film of water which runs down the inner containment wall surface. Provisions are made to collect and channel condensate to the IRWST, replenishing the steam losses and allowing the passive heat removal process to continue.

[1] The PRHR is also effective in removing heat from the RCS during a LOCA until voiding begins in the hot leg.

7. Figure 1 shows a diagram of the **AP1000** PCS.

8. In GDA Step 4, the various Structures, Systems and Components (SSCs) of the PXS and the supporting analyses were assessed in detail by ONR, including in the fault studies topic area (Ref. 1). For many aspects of the PXS, ONR found that the claims made by Westinghouse were supported by adequate analyses and experimental data. However, ONR was unable to find any substantiation in the European Design Control Document (EDCD) (Ref. 2) for the containment performance assumed for intact circuit faults. Significantly, there was no detailed justification provided for how much of the condensate forming on surfaces within the containment would be returned to the IRWST. Ref. 2 also stated that for intact circuit faults, the PXS would be capable of removing decay heat from the RCS indefinitely. To achieve this, high efficiency is required from the PXS but it is unavoidable that not all the condensate will be returned to the IRWST. Some of the steam will condense and get trapped on other structures within the containment. Alternatively, it could drain to the containment sump, bypassing the IRWST. Over a period of time (that time being dependent on how big the condensate losses are), the PRHR heat exchanger could uncover and cease to be effective.

9. Figure 2 illustrates the steam / condensate cycle, showing where losses can occur.

10. As a result of this lack of justification, at the end of GDA Step 4 ONR wrote the GDA Issue GI-AP1000-FS-06 requiring Westinghouse to provide validation that the IRWST is functionally capable of cooling the PRHR during intact circuit faults for 72 hours, or propose a design change to rectify the situation (Ref. 3).

1.3 Scope

11. The scope of this assessment is detailed in the assessment plan (Ref. 4). Consistent with this plan, the assessment is focused on considering whether Westinghouse's submissions to ONR for GI-AP1000-FS-06 provide an adequate response to justify the closure of the GDA Issue. As such, this report only presents the assessment undertaken as part of the resolution of the GDA Issue and it is recommended that this report be read in conjunction with the Step 4 Fault Studies – Design Basis Faults Assessment of the Westinghouse AP1000 Reactor (Ref. 1) in order to appreciate the totality of the assessment of the evidence on design basis reactor faults and the effectiveness of the PXS. As will be explained, the work that Westinghouse has undertaken for this issue has been extensive, including:

- a review of applicable phenomena
- experimental work of test rigs
- changes to the **AP1000** design
- revisions to analysis methodologies
- reanalysis of limiting design basis faults
- revisions to safety case claims and arguments

12. It has been necessary to consider all of these aspects as part of this assessment to come to a judgement on whether this GDA Issue on a fundamental aspect of the **AP1000** design can be closed.

1.4 Method

13. This assessment has been undertaken consistent with internal guidance on the mechanics of assessment within ONR (Ref. 5).

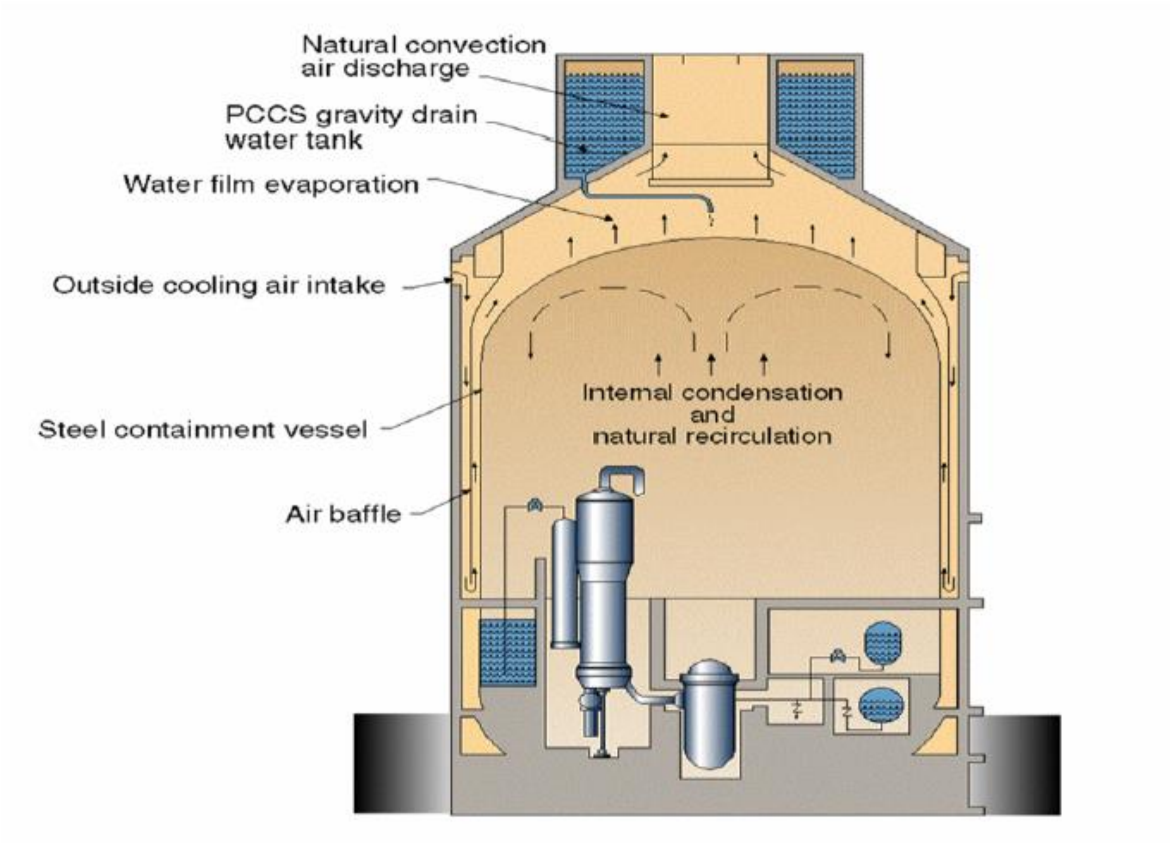

Figure 1: Overview of AP1000 PCS

1. Steam leaves IRWST
2. Pressurises containment
3. Condenses on walls, floors, structures
4. Condenses on containment vessel
   a. Most collected and returned to IRWST
   b. Some splashes / spills off
5. Losses from IRWST collect under reactor vessel

5a. In contact with the hot reactor vessel, steam rises up into containment

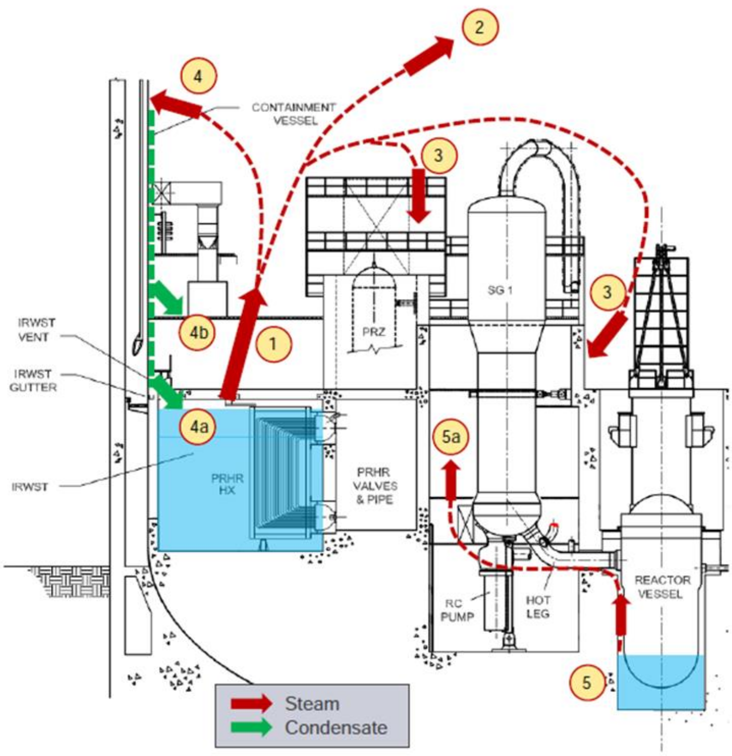

Figure 2: Steam / condensate cycle within the containment

2 ASSESSMENT STRATEGY

2.1 Pre-Construction Safety Report (PCSR)

14. ONR's GDA Guidance to Requesting Parties (Ref. 6) states that the information required for GDA may be in the form of a PCSR, and the Technical Assessment Guide NS-TAST-GD-051 sets out regulatory expectations for a PCSR (Ref. 7).

15. At the end of Step 4, ONR and the Environment Agency raised GDA Issue GI-AP1000-CC-02 (Ref. 8) requiring that Westinghouse submit a consolidated PCSR and associated references to provide the claims, arguments and evidence to substantiate the adequacy of the **AP1000** design reference point.

16. A separate regulatory assessment report is provided to consider the adequacy of the PCSR and closure of GDA Issue GI-AP1000-CC-02, and therefore this report does not attempt to assess the totality of the **AP1000** PCSR chapters related to fault studies (Chapters 8 and 9). However, there are some significant refinements to the claims made on the PXS and how a safe shutdown state can be demonstrated following design basis events which need to be appropriately captured in the PCSR. Therefore, as part of this assessment, I have reviewed and commented on the adequacy of the relevant revisions to the PCSR.

2.2 Standards and Criteria

17. The assessment has been undertaken in line with the requirements of the HOW2 BMS document NS-PER-GD-014 (Ref. 9). In addition, the Safety Assessment Principles (SAPs) for Nuclear Facilities constitute the regulatory principles against which dutyholders' safety cases are judged, and, therefore, they are the basis for ONR's nuclear safety assessment. The SAPs 2014 Edition Revision 0 (Ref. 10) has been used when performing the assessment described in this report (the original GDA Step 4 fault studies assessment used the 2006 Edition).

2.2.1 Safety Assessment Principles and Technical Assessment Guides

18. The following SAPs (Ref. 10) were identified in the assessment plan (Ref. 4) as being appropriate to judge the adequacy of the arguments in the area of fault studies for the UK **AP1000** reactor.

- Fault Analysis SAPs FA.1 to FA.9
- Severe Accidents SAPs FA.15 and FA.16
- Engineering SAPs EKP.2 to EKP.5, ECS.1, ECS.2, EDR.1 to EDR.4, ESS.2, ESS.4, ESS.6 to ESS.9, ESS.11, ERC.1 to ERC.3, EHT.1 to EHT.4
- Computer Codes and Calculation Methods SAPs AV.1 to AV.8
- Numerical Target for DBA Consequences Target 4

19. It is important to note, however, that the scope of the assessment to close out the GDA Issue is narrowly defined and is less than that of a typical ONR assessment, such as that undertaken in GDA Step 4. The original fault studies assessment (Ref. 1), which resulted in GI-AP1000-FS-06, considered the SAPs identified above. The objective of this assessment is primarily to judge the adequacy with which Westinghouse's submissions address the requirements of the GDA Issue, rather than repeat the original assessment against the SAPs.

20. A key expectation for this assessment is established by SAP FA.8 (paragraph 641) :

> The safety measures should be shown to be capable of bringing the facility to a stable, safe state following any design basis fault. Consideration should

therefore be given to the mission times required of SSCs when defining the performance requirements for delivering their safety functions.

21. The SAPs, which establish expectations for the assurance of validation of data and models, are important for this assessment, notably:

- AV.1: Theoretical models should adequately represent the facility and site.
- AV.2: Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.
- AV.3: The data used in the analysis of aspects of plant performance with safety significance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means.
- AV.6: Studies should be carried out to determine the sensitivity of the analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation.

2.2.2 National and International Standards and Guidance

22. There are both International Atomic Energy Agency (IAEA) standards (Ref. 11) and Western European Nuclear Regulators Association (WENRA) reference levels (Ref. 12) that are relevant to the fault studies assessment of the **AP1000** reactor. The original GDA fault studies assessment undertaken during Steps 3 and 4 took cognisance of the international standards published at the time. The GDA Issues that emerged from that original assessment can generally be characterised as having their origins in the application of the SAPs and UK-relevant good practice rather than through comparison with international guidance. Therefore, the SAPs (and not the international references) are the foremost standards considered. It should be noted that the latest version of the SAPs (Ref. 10) were benchmarked against the extant IAEA and WENRA guidance in 2014.

23. The IAEA has published a specific safety guide on deterministic safety analysis for nuclear power plants (Ref. 13). This provides recommendations on computer modelling of thermal hydraulic phenomena such as those considered by this GDA Issue and has direct relevance to several aspects. However, its requirements are consistent (although slightly more detailed) with the expectations set out in the fault analysis series of SAPs.

24. The **AP1000** reactor has been developed principally in the US with the aim of demonstrating compliance with US regulatory requirements. Similarly, the work to address the underlying technical challenges identified by GI-AP1000-FS-06 has been heavily influenced by the need to meet US regulatory requirements. Regulatory Guide 1.203 (Ref. 14) sets out what the United States Nuclear Regulatory Commission (US NRC) considers acceptable for developing and assessing evaluation models used to analyse design basis accidents. It is more detailed that either the SAPs or Ref. 13, and provides a comprehensive example of relevant good practice for transient analysis. For a new methodology being developed in the US, I would expect its advice to be followed.

2.3 Use of Technical Support Contractors (TSCs)

25. It is usual in GDA for ONR to use technical support, for example to provide additional capacity to optimise the assessment process, to enable access to independent advice and experience, analysis techniques and models, and to enable ONR‘s inspectors to focus on regulatory decision-making, etc.

26. As part of the assessment of GI-AP1000-FS-06, a contract was placed with Amec Foster Wheeler for it to review the details of Westinghouse’s analysis and the proposed design modifications, and to provide advice on their adequacy to ONR. The

report written by Amec Foster Wheeler (Ref. 15) was provided to Westinghouse before it (Westinghouse) issued its final submissions, allowing it to respond to some of the TSC's observations.

27. Ultimately, my assessment has been made against several key reports updated by Westinghouse after Amec Foster Wheeler's contract scope was complete. However, Ref. 15 has been a significant input to my regulatory judgement on the adequacy of Westinghouse's submissions.

2.4 Interaction with Overseas Regulators

28. Although GI-AP1000-FS-06 arose from the UK-specific GDA process and its closure is a matter for ONR, the underlying technical concern of a lack of substantiation of the condensate return losses was equally applicable to the 'standard' **AP1000** plants being built in the US and China.

29. At the end of GDA Step 4, ONR shared its assessment conclusions with its counterparts in US NRC and the (Chinese) National Nuclear Safety Administration (NNSA). Both they and Westinghouse recognised that the concern existed beyond the UK and even though regulatory interactions were suspended in the UK, Westinghouse initiated a significant workstream to address the issue.

30. At the point at which Westinghouse returned to the UK, major submissions, analysis methodologies and design changes had been submitted by Westinghouse to both US NRC and NNSA for evaluation. In the case of the US, the concern was also subject to scrutiny by the Advisory Committee on Reactor Safeguards (ACRS), which provides independent oversight and advice to US NRC.[2]

31. ONR is a member of the Organisation for Economic Co-operation and Development (OECD) facilitated Multinational Design Evaluation Programme (MDEP) **AP1000** working group. At the biannual meetings, ONR and other nuclear safety regulators considering the **AP1000** design (notably US NRC and NNSA) share and discuss issues of common interest. It became apparent, through both the discussions at these meetings and from dialogue with Westinghouse, that very similar (in many cases, identical) analyses and design changes were being submitted to the represented national regulators on similar timescales.

32. Each regulator has come to its own independent conclusion on the adequacy of Westinghouse's submissions, in accordance with its own regulatory and legal framework. However, MDEP **AP1000** working group meetings were used as a forum for keeping overseas colleagues informed about progress, emerging issues and ultimately regulatory conclusions. US NRC demonstrably undertook a rigorous review of Westinghouse's analysis, requiring Westinghouse to provide additional evidence and modify its analysis methodology (all before a final submission was provided to ONR). It also performed its own independent confirmatory modelling of condensate recirculation to support its judgements. In turn, the ACRS scrutinised both Westinghouse's and US NRC's evaluations.

33. US NRC's final safety evaluation report summarising its regulatory assessment of the condensate return issue (Ref. 16) and the ACRS letter recommending acceptance of analysis and design changes (Ref. 17) are publicly available. I have taken cognisance of many aspects of this report and its findings in my own assessment. In a similar way to how I have used the TSC work (Ref. 15), familiarity and access to this overseas

[2] US NRC did not deal directly with Westinghouse. Instead, it received submissions from one of the applicants planning to build **AP1000** plants (Duke Energy who plan to build two units at the Levy site) to change the **AP1000** design it had certified under Title 10 of the Code of Federal Regulations Part 52. However, it was Westinghouse that performed the analysis under the control of the future licensee.

assessment of the same issue has allowed me to target my attention at the UK-specific safety case arguments.

2.5 Integration with Other Assessment Topics

34. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot therefore generally be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. However, in the case of this GDA Issue, almost all of the areas under consideration are associated with thermal hydraulic modelling and changes to the fault studies sections of the PCSR (Chapter 9). Therefore, colleagues in other technical disciplines have had limited involvement in this assessment.

35. An outcome of Westinghouse's work to address this GDA Issue has been some design changes to the water collection / return system and to components attached to the side of the containment wall which could interfere with the water collection / return system (notably the polar crane girder and the containment stiffener). I was advised on the adequacy of these changes by mechanical engineering specialists as part of the Amec Foster Wheeler contract.

2.6 Out of Scope Items

36. At the time of drafting GI-AP1000-FS-06, ONR had an expectation for the scope of work required to provide adequate validation evidence for the effectiveness of the PRHR / IRWST / PCS following an intact circuit fault. The scope and breadth of work Westinghouse needed to undertake to fully address this GDA Issue (and the requirements of other regulators) ultimately exceeded these initial expectations. However, none of the submissions provided by Westinghouse have been excluded from the scope of this assessment (although, as stated above, in some cases credit has been taken for assessment work done by Amec Foster Wheeler and US NRC to ensure that my time and regulatory attention could be appropriately targeted).

3 REQUESTING PARTY'S DELIVERABLES IN RESPONSE TO THE GDA ISSUE

37. Westinghouse's principal submission is Ref. 18. This is a UK-specific report written to address GI-AP1000-FS-06 and the requirements of the UK **AP1000** safety case. However, it summarises several years of work that is generic (ie applies to all **AP1000** plants) and includes references to many detailed reports and calculations that have been submitted to nuclear safety regulators in the US and China.

38. Ref. 18 describes the following:

- Work to identify and quantify the important phenomena that influence the condensate return rate to the IRWST, assembled in the form of a Phenomena Identification and Ranking Table (PIRT).
- An initial series of tests ('Phase 1') to investigate the behaviour of a condensate film as it flows down the vertical side wall of the containment shell. This had been identified in the PIRT as a phenomenon with a high importance but a low state of knowledge.
- Following a second PIRT, a second series of tests ('Phase 2') performed with a different experimental rig to more accurately simulate other important phenomena assuming prototypic conditions.
- Design changes (identified following the test work) to the polar crane girder, internal stiffener and IRWST gutter, as well as the addition of a downspout piping system, were made to facilitate water collection; additionally a reduction in the number of PRHR heat exchanger tubes it is permissible to plug (down from 8% to 5%) was made.
- A new analysis methodology developed specifically to consider the condensate return issue. The methodology is made up of the following:
    - Containment response analysis using the Westinghouse GOTHIC (WGOTHIC) code which tracks the condensation that bypasses the PXS gutter arrangement.
    - Hand calculations which evaluate the overall percentage of steam condensation that is lost from the containment vessel shell (used to justify the basis for a bounding containment vessel shell bypass as an input into the containment response analysis).
    - Analyses using the LOFTRAN code to evaluate the RCS cooldown following the PRHR heat exchanger operation with three objectives: a) to demonstrate the capability of the PRHR heat exchanger to cool the RCS core average temperature to 215.6°C (420°F) in 36 hours on a realistic basis, b) to demonstrate that conservative design basis analyses of events considered in Chapter 15 of the EDCD (Ref. 2) can be extended out to 72 hours with all safety criteria met, and c) to demonstrate that the PRHR heat exchanger can effectively match the decay heat and keep the RCS temperature below 215.6°C (420°F) for an extended period of time (eventually claimed to be at least 14 days).
    - Analyses using the RELAP code to independently confirm that the conclusions reached from the LOFTRAN calculations are appropriate despite two known simplifications: a) LOFTRAN neglects ambient heat losses to maximise the RCS energy, and b) LOFTRAN has a limited capability to model two-phase flow and therefore the performance of the PRHR when sub-cooling is lost.
- Recommended changes to the UK **AP1000** PCSR as a result of this generic work.

39. All the major aspects listed above are supported by additional references clearly identified in Ref. 18.

40. It should be noted that Ref. 18 describes the final methodology and references the last calculation results developed by Westinghouse. Earlier versions were initially

submitted to other regulators and changes were made in response to challenges and learning gained through these interactions. Similarly, ONR's TSC reviewed an early version of Ref. 18 and did not have access to some of the later references. Westinghouse's objective for the final version of Ref. 18 was that it would address the comments raised by the TSC as appropriate.

41. Westinghouse has incorporated its work in response to this GDA Issue into the UK **AP1000** safety case through changes to Chapter 9 of the PCSR (Ref. 19).[3] The design basis faults originally considered in Chapter 15 of the EDCD (Ref. 2) are captured in the main text of Chapter 9. The analysis for these faults (especially for the non-LOCA faults) remains focused on demonstrating a margin to safety criteria in the crucial first few seconds and minutes following an initiating event, and therefore has not been modified as a result of the work to address GI-AP1000-FS-06 (other changes have been made compared with what was included in Chapter 15 of the EDCD, and these are discussed further in the ONR assessment of GI-AP1000-FS-02, Ref. 20).

42. The discussion and demonstration of the UK **AP1000** reactor's ability to achieve a stable, safe shutdown state following design basis faults is included in Appendix 9C of the PCSR Chapter 9 (Ref. 19). Alongside discussion on demonstrating an ability to achieve a stable, safe shutdown state for LOCA faults and using lower safety class active SSCs, pertinent to this GDA Issue Westinghouse makes the following claims (supported by references):

- The Class 1 PRHR heat exchanger, Core Makeup Tanks (CMTs) and PCS can be shown by analysis (making conservative bounding assumptions) to be able to maintain the plant in a stable, safe shutdown state for at least 72 hours.
- Based on less conservative analysis, the PRHR heat exchanger is capable of reducing the RCS temperature to below 215.6°C (420°F) within 36 hours and can maintain this temperature for longer than 14 days.

43. Both of these claims are subtly, but importantly, changed from those originally presented in the EDCD (Ref. 2) and assessed during GDA Step 4 (Ref. 1). The first change is that Westinghouse had previously defined the safe shutdown state as being below 420°F (215.6°C). However, after crediting the identified design changes and using the revised methodology set out in Ref. 18, Westinghouse has not been able to show that these temperatures could be reached and maintained for 72 hours if conservative analysis assumptions (consistent with 'traditional' design basis analysis expectations) were made. However, Westinghouse's assertion is that the conditions that can be achieved by the passive Class 1 SSCs do represent a stable, safe shutdown state, specifically:

- the average RCS temperature will be less than 'no-load' (after the PRHR heat exchanger has matched the decay heat);[4]
- the RCS pressure will be less than the declared safety limit;
- the pressuriser water volume will be less than full;
- the steam generator pressure will be less than the declared safety limit; and
- the Departure from Nucleate Boiling Ratio (DNBR) will be greater than the declared safety limit.

44. The second change is to how long the PRHR heat exchanger is capable of maintaining a safe shutdown state. There were multiple statements in the EDCD (Ref. 2) that the

[3] The final wording of the safety case claims and arguments included in PCSR Chapter 9 was broadly consistent but not exactly as originally anticipated and recommended by Ref. 18.
[4] Westinghouse has defined 'no-load' as the normal at-power operating temperature and pressure (typically means 15.51 MPa abs [2250 psia], 292°C [557 °F]) with the reactor critical but producing no power. If this temperature is exceeded, the emergency procedures will prompt to initiate open loop cooling.

PRHR heat exchanger (in conjunction with the PCS) can keep the reactor in the declared safe shutdown state for an indefinite period of time. Following the work set out in Ref. 18, Westinghouse has qualified this claim to be “longer than 14 days”. Westinghouse supports this claim with ‘realistic’ analysis showing that the PRHR heat exchanger is able to match the decay heat for at least this long and that the plant is stable. The time constraint has been introduced because Westinghouse recognises that even with the design changes to condensate collection systems, ‘lost’ inventory will result in a decrease in the IRWST level and a degradation of the PRHR efficiency. As the PRHR is uncovered, the driving head within the tubes is reduced and the PRHR flow rate decreases. Eventually, the PRHR heat transfer rate will begin to lag behind the decay heat generation, resulting in an increase in core average temperature. However, as long as the bottom horizontal section of the PRHR remains covered, Westinghouse states that it expects this increase to be gradual in nature.

45. Westinghouse’s analysis does not show a cliff-edge after 14 days. At 14 days, although a significant portion of the PRHR will be uncovered, what is underwater is shown to be able to match the decay heat and maintain a relatively steady temperature at around 200°C (Ref. 21). The analysis suggests that it would take over 28 days for the PRHR to be fully uncovered and effective RCS cooling to be lost.

46. When Westinghouse extended out its conservative analysis of the intact circuit transient beyond 72 hours, it found it would take about seven days for the effectiveness of the PRHR to reduce sufficiently for the ‘no-load’ temperature limit defined in emergency procedures for actuating open loop cooling to be reached. It states that once the plant reaches an open loop cooling state, core cooling can be maintained for an indefinite period of time, as demonstrate by its analysis of LOCA faults (Ref. 19).

4 ONR ASSESSMENT OF GDA ISSUE GI-AP1000-FS-06

47. In the following subsections I have summarised my assessment of Westinghouse's submissions for this GDA Issue.

48. I have broken up my assessment into a number of key areas:

- use of PIRTs and testing
- physical design changes to the plant
- containment and condensate modelling
- design basis accident analysis modelling
- TSC observations on the selection of the limiting transient
- definition and demonstration of a stable, safe state
- adequacy of the PCSR

49. In all cases, the judgements that are presented are my own. However, in many of the areas considered I am relying on, or giving significant credence to, the detailed assessment work done by others, in particular Amec Foster Wheeler (under a contract placed to support this ONR assessment) and US NRC.

50. Ultimately, the basis for closing the GDA Issue is the adequacy of the documents submitted by Westinghouse. However, my own understanding of the technical challenges and Westinghouse's appreciation of ONR's requirements have been aided by multiple meetings and correspondence with Westinghouse over many months as it first developed and then finalised its methodologies, design changes and documentation.

4.1 Use of PIRTs and Testing

51. During GDA Steps 3 and 4, ONR's fault studies assessors (supported by TSCs) undertook a detailed assessment of Westinghouse's work to demonstrate the effectiveness of the **AP1000** reactor's novel passive systems (Ref. 1). Generally, ONR found that adequate demonstrations had been provided (with the exception of the matters identified in this GDA Issue). Key to reaching that judgement was the extensive use Westinghouse has made of experimental test facilities to understand how the proposed SSCs are expected to perform and to validate the computer models used to predict the overall **AP1000** system performance in fault conditions. A notable aspect of this original validation work was the extensive and systematic use of PIRT methods to determine the requirements for physical model development, scalability, validation, and sensitivity studies.

52. The application of PIRT approaches is widely recognised as relevant good practice (Ref. 14). I was therefore pleased to see in Ref. 18 that Westinghouse has continued to apply the rigour and insights of this approach at the start of its work to address this GDA Issue, and then again on several subsequent occasions to review its model development and design change efforts.

53. Westinghouse's initial PIRT to identify phenomena important to the condensate return losses and gaps in knowledge (Ref. 22) was reviewed by Amec Foster Wheeler's technical specialists on ONR's behalf (Ref. 15) and was found to be adequate. Amec Foster Wheeler's judgement was that Westinghouse had identified all the relevant phenomena and it agreed with Westinghouse's conclusions about which of these phenomena were well understood and which required further investigation.

54. This initial PIRT led directly to the Phase 1 tests to investigate the behaviour of a condensate film as it flows down the vertical side wall of the containment shell. Ref. 18 states that three different configurations of the test rig were used for differing objectives:

- Configuration 1 modelled the section of the containment vessel wall from the internal stiffener down to the IRWST gutter. This test quantified how much condensation was lost at the stiffener and the flow that was not captured by the IRWST gutter.
- Configuration 2 modelled the upper portion of the containment vessel wall from the polar crane girder down to the internal stiffener. The objective of this configuration was to determine the amount of condensate captured at the stiffener.
- Configuration 3 modelled a portion of the containment vessel wall where flow interacts with an attachment plate.[5] The objective of this configuration was to test the various geometries and losses associated with the attachment plates welded to the containment vessel wall.

55. The insights from this testing resulted in several design changes being made to the plant that will be discussed in the next subsection and informed the assumptions made about losses from obstacles in the new analysis methodology.

56. The containment dome is made up of plates of steel welded at four heights between the vertical and horizontal. Westinghouse's methodology had always allowed for a 'rain-out' phenomenon from shallow portions of the containment dome as a likely loss mechanism (see Figure 3).

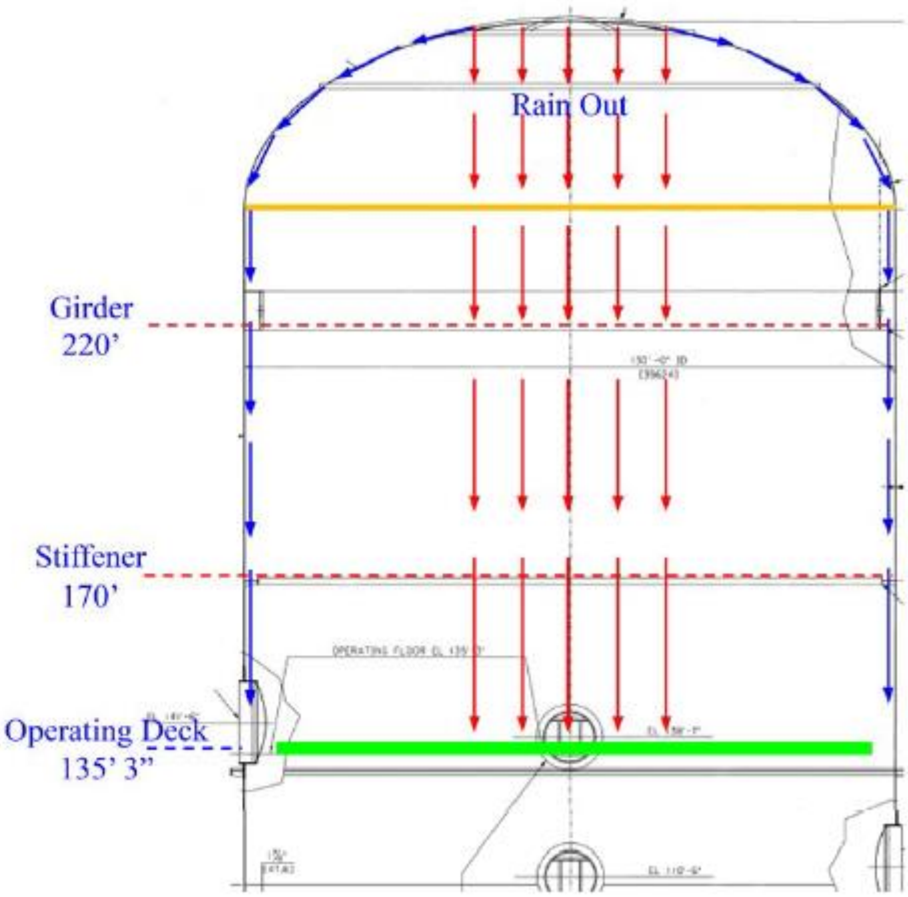

Figure 3: Rain-out phenomenon

[5] There are a large number of plates welded onto the inner wall of the containment vessel put there to provide anchorage for piping, HVAC (heating, ventilation and air conditioning) ducting, conduit, etc. The plates provide an obstacle to the condensate flowing down the inner wall and are a source of losses to the IRWST.

57. Westinghouse's modelling assumption is that any condensate passing the higher weld seams will be lost but condensate passing over the lower weld seams will be retained. However, Amec Foster Wheeler commented in Ref. 15 that it was not convinced by the recommendation made by the original PIRT (Ref. 22) to use of literature or bounding estimates to quantify this loss mechanism. It also raised a concern that the ASME design code defining the requirements for the containment vessel permits misalignments between the welded plates of the dome (within prescribed limits). While recognising that this issue was captured in the PIRT, Amec Foster Wheeler used hand calculations assuming limiting weld and plate geometries to cast doubt on Westinghouse's assumptions that the angle experienced by the condensate film at the lower seam could be steep enough for rain-out to be disregarded.

58. A third concern raised by Amec Foster Wheeler was the temperature dependence revealed by Westinghouse's Phase 1 testing of the condensate losses due to obstacles (Configuration 3). While Amec Foster Wheeler agreed with Westinghouse that there was a temperature dependence, it was not satisfied that the findings of the Phase 1 tests (which used non-prototypic temperatures) could be applied to the conditions that would be seen inside an **AP1000** containment during an extended intact circuit transient with an adequate level of confidence.

59. I supplied Ref. 15 to Westinghouse, and asked it to respond through a regulatory query (Ref. 23) to the observations made by Amec Foster Wheeler. As a result, Westinghouse updated Ref. 18 with details of a second PIRT undertaken after the Phase 1 test programme was completed. Westinghouse explained that through its second PIRT it had identified itself many of the limitations flagged by Amec Foster Wheeler about the Phase 1 test. This had prompted it to initiate its Phase 2 programme using a new test facility to simulate:

- prototypic containment temperature, pressure, and steam / air fraction
- prototypic condensate flow rates
- prototypic heat transfer rate through the containment wall
- water losses associated with angle of orientation of contact surface, including impact of discontinuities (welds, plate alignments) on containment vessel dome

60. Westinghouse's notable conclusions from the Phase 2 programme are as follows:

- The transition angle from rain-out to film flow is at an angle that supports the existing analysis.
- The higher weld seam angles cause a large loss of condensate flow.
- The intermediate weld seam angles cause a minor loss of condensate flow.
- The lower weld seam angles do not result in any loss of condensate flow.
- Attachment plates welded to the containment dome at steep inclination angles will strip off all the condensate flowing over the vessel.
- Attachment plates welded to the containment dome between the steepest region and the vertical will result in losses that are mostly associated with the width of the beam attached to the plate.
- Attachment plates welded to the vertical portion of the containment vessel will cause virtually no losses.

61. On this basis, Westinghouse argues that its original modelling assumptions developed from the Phase 1 testing are conservative and do not need to be changed.

62. I am satisfied that, through this second phase of testing, Westinghouse has been able to robustly address all of Amec Foster Wheeler's challenges on the adequacy of the condensate loss assumptions. It should be noted that Westinghouse has stated that it originally undertook the Phase 2 work to investigate and quantify margins for potential future applications. However, because it ultimately supported the original analysis assumptions, it had not formally included the Phase 2 work in its submissions to other

regulators. From a UK perspective, I consider the Phase 2 test programme is important to closing out this GDA Issue, and its inclusion in Ref. 18 has strengthened Westinghouse's safety case.

63. I established through discussions with Westinghouse and my interactions with other regulators at MDEP **AP1000** working group meetings that, at the two sites in China where **AP1000** units are being constructed, manual welding techniques have been used to join the plates of the containment vessel. At US **AP1000** construction sites, automated welding techniques have been deployed. I have been assured that in both countries the results have been consistent with the required codes governing the adequacy of the welds; but the automatic techniques have resulted in uniform welds, while in China (for various reasons) it has been necessary to grind down the seams. In response to a regulatory query (Ref. 23), Westinghouse stated that its Phase 2 tests assumed the maximum misalignments between plates and worst orientations relative to the flowing water film. It does not expect any weld on a UK **AP1000** containment to be worse than that assumed in the testing and as a result this GDA Issue does not result in any additional requirements for which welding techniques should be used, or establish a requirement to grind weld seams flat. I have no reason to doubt this conclusion. As with all safety case assumptions for new builds, the constructed plant (in this case, the welds) will need to be shown to be consistent with the safety case, or the plant and / or safety case will need to be modified. This will be part of 'normal business' for a future licensee to demonstrate during construction.

64. Ultimately, I am satisfied with how Westinghouse has addressed this aspect of the GDA Issue's requirements. Its use of PIRTs to inform the testing and modelling is consistent with relevant good practice (eg as established by Ref. 14). Its use of physical testing to support its analysis methods is consistent with SAP AV.2 (Ref. 10) and is well documented (including good descriptions and photos of the test facilities) in accordance with SAP AV.5. Amec Foster Wheeler's detailed assessment did find some gaps in the early submissions but I consider the updated version of Ref. 18 to be adequate in addressing these concerns.

65. In addition to the deployments of PIRT techniques discussed above, Westinghouse references a third application of PIRTs in Ref. 18 to identify and rank the phenomena important to long-term PRHR heat exchanger operation (Ref. 24).This has informed Westinghouse's understanding on the limitations of LOFTRAN and prompted it to validate its predictions against RELAP analysis (an alternative thermal hydraulic code with a greater capability to model two-phase flow) and the results of physical testing. I will discuss this further in Subsections 4.4 and 4.6 below. However, I again welcome the sensible use Westinghouse has made of PIRT techniques to inform its analysis.

4.2 Physical Design Changes to the Plant

66. Ref. 18 describes and references several design changes to the **AP1000** design (affecting all proposed plants worldwide, not just the UK **AP1000** plants) resulting from the insights gained from the testing.

67. The polar crane girder is made up of 80 boxes which are welded together around the circumference of the containment vessel. At each interface where the sections are welded together, all four corners have openings to prevent multiple welds from joining at a common location. Therefore, as part of the original design intent, each of the 80 box sections would have been open at the corners. The internal stiffener also uses this design. It was envisaged that these fabrication holes at the corners would allow the condensate to flow past the two major obstructions and continue onto the IRWST gutter. However, when tested, this was found not to be happening to a sufficient extent. Design Change Proposal (DCP) APP-GW-GEE-3692 (Ref. 25) sets out the following changes to the design to address this problem:

- All the fabrication holes on the top surface of the polar crane girder and in the stiffener are to be blocked with ¼ inch-thick, half-circle plates to prevent flow from passing through these areas.
- A downspout piping network is to be added to collect and transport condensation from the polar crane girder and stiffener to the IRWST gutter collection boxes. This piping network is Class 1 and seismic category I.
- Screens similar to those already included on the IRWST gutter design are to be added to the entrance of each of the downspouts at the polar crane girder and stiffener, to prevent any larger debris from blocking the downspout piping.

68. The original design of the IRWST gutter included a small drip lip immediately above it attached to a continuous support plate running around the containment. Informed by the test programmes, Ref. 25 adjusts the angle of the drip lip and extends its length to minimise losses to the IRWST gutter. It also modifies the IRWST gutter arrangement to have gutters above the personnel airlock and equipment hatch which connects main portions of the gutter installed at a lower level.

69. DCP APP-GW-GEE-4657 (Ref. 26) was written after Ref. 25 following a Westinghouse design review of its original proposals. It was realised that the original downspout proposal only collected water from the top surface of the polar crane girder boxes, neglecting the condensation which would form and collect inside the boxes. Ref. 26 addresses this by a) sealing the bottom of the polar crane girder boxes, b) adding flow communication holes between the boxes, and c) providing four additional downspouts to the bottom of the boxes that tie into the downspout architecture proposed by Ref. 25.

70. Westinghouse states in Ref. 18 that all these identified modifications are directly related to improving the efficiency of the return of condensation from the containment vessel shell, and they extend the time for which the PRHR heat exchanger will remain effective. On that basis, I clearly welcome these changes. Given their nature and the fact that they have been identified early in the UK **AP1000** project (before construction has started or even contracts placed), they are likely to add little to the overall cost of the plant and be straightforward to install, while making a significant contribution to nuclear safety. They are therefore demonstrably reasonable, practicable modifications that help to reduce risks (ie they are consistent with the UK legal requirements of demonstrating that the design reduces the risk As Low As Reasonably Practicable (ALARP)).

71. I have examined the latest version of the design reference point report for the UK **AP1000** plant (Ref. 27) and found both DCPs included in Table 7 (DCPs written during the GDA Issue closure phase, which Westinghouse states meet its highest criteria for safety significance and will be credited in the relevant sections of the PCSR). Given my comments above on the ALARP nature of these changes, I welcome their inclusion in Ref. 27.

72. As part of its detailed review on behalf of ONR, Amec Foster Wheeler reviewed the design changes proposed by Westinghouse (Ref. 15). It did its own calculations to check that the gutters and downpipes would have sufficient capacity for the expected flows under postulated conditions (including an assumption that one of the principal downspouts was blocked) and reported that it was satisfied with Westinghouse's proposals. It also looked at the proposed drip lip arrangements and challenged Westinghouse through a regulatory query on whether it had chosen the optimum solution based on the Phase 1 test programme. Both Amec Foster Wheeler and I were satisfied with the response provided by Westinghouse (Ref. 28).

73. I note that the same or very similar physical changes have been proposed to (and accepted by) the relevant regulators in the US and China. The exact implementation has been slightly different in China because the affected parts of the design had

already been installed. In both countries, the cost and difficulty of making the changes were more significant than they will be in the UK because the placing of contracts for components and progress with construction were significantly more advanced. However, this extra cost and difficulty did not stop Westinghouse proposing the changes to its customers or the relevant regulators accepting them.

74. Ref. 18 identifies two other DCPs produced as a result of the testing programme (Refs 29 and 30). However, they are not proposing physical modifications. Instead, they are acting as vehicles to initiate the necessary changes to the analysis methods and safety case claims made in the affected licensing documentation across the different countries building the **AP1000** units. Both DCPs are included in Table 7 of Ref. 27 for inclusion within the GDA design reference point and safety case. Therefore, I am assuming that by assessing the analysis described in Ref. 18 and the modified PCSR (Ref. 19), I am effectively commenting on the adequacy of these two DCPs.

4.3 Containment and Condensate Modelling

75. Westinghouse's methodology for modelling the containment response and the amount of condensate that bypasses the PXS gutter arrangement has been assessed in detail by both Amec Foster Wheeler (Ref. 15) and US NRC (Ref. 16).

76. Ref. 15 makes no significant negative observations on the adequacy of Westinghouse's WGOTHIC modelling and its hand calculations. It does observe that the assumptions made in Westinghouse's methodology have been chosen to maximise the impact on the IRWST water level in the long term but Amec Foster Wheeler's assessors had concerns that pressuriser level may be a safety criterion of concern for some transients. For such transients, Amec Foster Wheeler suggested that assumptions that maximise containment pressure would be more limiting than those assumed by Westinghouse. I will return to these observations in Subsection 4.5.

77. Ref. 16 describes in some detail US NRC's lines of regulatory enquiry, the responses provided by the relevant 'applicant' (the prospective operator of the lead US **AP1000** plant presenting Westinghouse's work to US NRC) and US NRC's ultimate conclusions. My own judgements are informed by the following from Ref. 16:

- US NRC examined submissions that describe the WGOTHIC model and how it calculates the containment pressure, containment temperature, and steaming rate from the IRWST to the containment atmosphere, heat sinks and the containment shell. It judged that these submissions provide an accurate summary of the analysis, including an explanation of how the containment response calculation relates to other calculations, the input assumptions, and the key results, with sufficient information for US NRC staff to reach its findings.

- US NRC considered in detail what happens to condensate that is lost from the containment vessel wall. The applicant stated that most of the condensate that is lost from the containment vessel will eventually reach the reactor cavity (see Figure 2). The water level here will rise until it reaches the reactor vessel, at which point steaming will begin and potentially add to the net condensate return fraction to the IRWST. Ref. 16 describes how US NRC secured sensitivity studies from Westinghouse (via the applicant) to gain an informed appreciation of the importance of reactor vessel steam. The applicant reported that if no steaming was accounted for, the IRWST water level would be reduced by 7 inches in a 72-hour period. This would have no appreciable impact on the PRHR heat exchanger performance over that period. US NRC performed its own sensitivity studies using both LOFTRAN and MELCOR, and found that its results were bounded by the applicant's. On that basis, US NRC concluded that Westinghouse's treatment of steaming from the reactor cavity was acceptable.

- US NRC considered the applicant's arguments and assumptions for condensation holdup on surfaces such as the operating deck and large pieces of equipment. The original approach was found to assume a horizontal film holdup volume proportional to the cross-sectional area of the containment with a multiplication factor applied. When US NRC investigated the justification for the applied factor, it came to the conclusion that it may not be conservative. It actioned Westinghouse (via the applicant) to undertake a sensitivity study using a different approach to determine the holdup surface area, and to use an alternative method for estimating the film thickness. Neither of these changes resulted in a significant impact on the performance of the PRHR heat exchanger during the first 72 hours of an intact circuit fault. An inspection by US NRC of the final Westinghouse documentation for containment response analysis with long-term PRHR operation confirmed that the revised methods were being used.

- US NRC considered the modelling of the general characteristics of the containment response during the transients of interest. Westinghouse's analysis found that containment pressure and temperature, IRWST temperature, and the ambient outside temperature all have an impact. Ref. 16 discusses how the pressure response can be divided into two phases: an initial spike in pressure as the IRWST boils off, followed by a slow levelling off to a peak and decay as passive cooling occurs. It goes on to say that US NRC's internal confirmatory analysis using MELCOR predicted a similar trend to that predicted by Westinghouse using WGOTHIC, with Westinghouse's pressure predictions bounding the MELCOR results at all points after the first hour of IRWST steaming. More significantly, Ref. 16 observes that Westinghouse's analysis predicts a higher saturation pressure for water in containment, which results in additional holdup in the containment atmosphere and higher IRWST temperatures, and therefore reduced heat transfer from the PRHR. It therefore concludes that Westinghouse's modelling of the containment pressure response is conservative.

- Ref. 16 details a dialogue between US NRC and the applicant / Westinghouse on the various initial temperature assumptions made. The maximum inside and outside containment temperatures considered by Westinghouse in its **AP1000** design basis safety case are 49°C and 46°C respectively. However, for these analyses, Westinghouse has assumed an initial in-containment temperature (including all heat sinks and the IRWST) of 29°C together with the maximum 46°C external temperature. Westinghouse took into account a number of competing effects to arrive at these assumptions. A lower internal temperature results in additional condensation on (internal) heat sinks but it also delays the time to boiling in the IRWST. Both Westinghouse and US NRC were in agreement that the lower heat sink temperature slightly outweighed the effect of the IRWST temperature. The influence of exterior temperature was found to be more significant on PRHR performance. Higher ambient temperatures result in higher initial PCCWST water temperatures, which in turn results in less heat removal from containment during a transient and thus higher containment pressures and temperatures. Westinghouse has performed a study for a plant located at a site where the ambient temperature could reach 46°C and calculated that the in-containment temperature during normal operation with containment coolers running would not be below 31°C. US NRC agreed that 29°C was therefore an appropriately conservative value to use.

78. Although I have not examined the supporting references identified in Ref. 18 to the same level of detail as Amec Foster Wheeler and US NRC, it is clear from a high-level review that Westinghouse has demonstrably documented its containment modelling analysis clearly and systematically, consistent with the expectations of SAP AV.5. On

the basis of what is set out in Ref. 16, I am also satisfied that appropriate sensitivity studies have been performed (in accordance with SAP AV.6) by Westinghouse and I take further reassurance from the independent evaluations performed by US NRC.

79. Finally, I am satisfied with the assumptions on initial temperatures assumed by Westinghouse. On the basis that the temperature of the PCCWST water on the outside of the containment is a dominating factor for heat removal, assuming an ambient outside temperature of 46°C should be conservative for any **AP1000** plant built in the UK. The heat sinks inside the containment have a large thermal inertia and should see a sizeable but steady heat load from a normally operating plant, and given small reported impact from competing effects, I am satisfied that the assumed in-containment pre-fault temperature is reasonable for a UK plant.

4.4 Design Basis Accident Analysis Modelling

80. For all **AP1000** plants, including in the UK, Westinghouse has analysed a standard list of design basis intact circuit faults using well-established methodologies and summarised in the work in Chapter 15 of the **AP1000** design control document (in the case of the UK, it was the EDCD, Ref. 2; going forward this information will be in Chapter 9 of the PCSR, Ref. 19). Through these analyses, Westinghouse demonstrates that a set of safety acceptance criteria are met, notably:

- minimum DNBR
- maximum RCS pressure
- maximum steam generator pressure
- maximum pressuriser water volume

81. However, for many of the considered faults, the challenging period for meeting the relevant acceptance criteria is in the first few seconds or minutes following the initiating event. Carrying on the transient analysis to model the plant behaviour beyond this period of interest was therefore seen to be of little benefit, even though the safety case argument was that all acceptance criteria would be met for 72 hours through the performance of the PXS. For a small number of non-LOCA events, the analyses were extended out for several hours but even the longest of these (inadvertent CMT operation at power) only went to circa 9 hours. As a result, condensate losses and PRHR heat exchanger uncovery were not being adequately considered. This was a significant observation for ONR at the end of GDA Step 4 which resulted in GI-AP1000-FS-06; a safety case claim was being made that the PXS would be able to cool the RCS for 72 hours following an intact circuit fault but little evidence had been supplied to support it.

82. Ref. 18 states that the limiting transient for removing core decay heat and demonstrating the effectiveness of the PXS is a loss of normal feedwater fault with a coincident loss of AC power. Westinghouse arrived at this conclusion after studying several variations of loss of feedwater faults, main steam line break faults and feedwater line breaks using the MAAP4 code (Ref. 31).

83. Westinghouse then reanalysed this bounding fault using an adapted version of the LOFTRAN code to demonstrate that all safety criteria are met out to 72 hours. The original LOFTRAN analyses in EDCD Chapter 15 (Ref. 2) assumed a constant 90% condensate return rate. The new modelling takes the time-dependent containment pressure, containment temperature, IRWST steaming rate and PXS condensate return flow rate and temperature from the containment analysis in the form of input tables. Conservative input parameters have been assumed (eg a decay heat with two sigma uncertainty included) consistent with Westinghouse’s normal design basis analysis methods.

84. Westinghouse states in Ref. 18, referencing results in Ref. 32, that this bounding analysis shows that no safety acceptance criteria are challenged during the full 72-hour transient. This conclusion and supporting results are also summarised in Appendix 9C of PCSR Chapter 9 (Ref. 19).

85. Both Amec Foster Wheeler and US NRC examined this analysis in detail.[6] In Ref. 15, Amec Foster Wheeler states that it was satisfied (with one exception) that the LOFTRAN analysis had used appropriately conservative assumptions and data for this design basis demonstration (ie the expectations of SAP FA.7 are being met). The one exception was the assumption on ambient heat losses from the primary circuit. This was also identified by US NRC and I will discuss this point later in this subsection. Amec Foster Wheeler also queried the choice of bounding fault made by Westinghouse. I have addressed this second point in Subsection 4.5 below.

86. In Ref. 16, US NRC discusses the following, which I have taken into account in my assessment:

- It agrees with Westinghouse that a loss of normal feedwater fault with a coincident loss of AC power is the limiting event because it combines a relatively late reactor trip with a significant loss of secondary-side inventory to both steam generators and the loss of forced reactor flow, resulting in the largest mismatch between primary-side energy and secondary-side / PRHR heat removal capability. It states that it performed its own confirmatory analysis to investigate the impact of the choice of initiating event. The conclusions of this work were consistent with Westinghouse's.

- It observes that when it had previously made positive judgements on the acceptability of LOFTRAN to analyse **AP1000** design basis transients, extended transients that experience uncovery of the PRHR heat exchanger tubes had not been considered. LOFTRAN assumes a collapsed liquid level in the IRWST, and any PRHR heat exchanger surface area above that collapsed water level is not credited for heat removal. US NRC was able to satisfy itself that this is appropriate, and that previously raised concerns about the potential for vapour blanketing from lower tubes to impede heat transfer from upper (covered) tubes would actually be less of a concern as the PRHR heat exchanger starts to uncover.

87. US NRC makes similar observations in Ref. 16 to those made by Amec Foster Wheeler in Ref. 15 on the assumptions about heat losses from the primary circuit. Westinghouse's 'normal' LOFTRAN analyses for intact circuit faults assume the RCS to be adiabatic, resulting in the highest required heat removal from the PRHR heat exchanger. However, a concern was identified that ambient heat losses from the RCS (in particular the pressuriser) would result in a lower RCS pressure to that assumed in the modelling and the margin to the point where subcooling is lost could be degraded. A loss of subcooling could result in steam forming at the high points of the system such as the top of the reactor vessel or the PRHR, potentially causing a reduction in PRHR heat exchanger performance.

88. Westinghouse addresses this point in Ref. 18 and its supporting references. It states that including these heat losses in long transients will result in a small reduction in RCS temperature but in a more significant reduction in RCS pressure. It also recognises that the LOFTRAN code has a very limited ability to model two-phase flow. However, it asserts that there is no change in PRHR performance resulting from the loss of subcooling. It supports this claim by referencing RELAP analysis and integral

[6] ONR assessed the adequacy of the LOFTRAN code for EDCD Chapter 15 faults during GDA Step 4 (Ref. 1). I have not attempted to repeat the assessment or challenge the positive conclusions reached in GDA Step 4 about the general use of LOFTRAN.

systems testing at the APEX facility (discussed in ONR's GDA Step 4 Fault Studies report, Ref. 1). Both the RELAP modelling and the test results show that the PRHR heat removal follows the decay heat for both subcooled and saturated RCS conditions.

89. Ref. 16 describes how US NRC audited Westinghouse's documentation detailing the modelling of ambient heat losses from the pressuriser and was satisfied with what it found. It also states that US NRC performed its own confirmatory calculations and obtained results consistent with Westinghouse's. It concluded that the treatment of ambient heat losses in the analysis of design basis transients (ie neglecting them) is suitably conservative. Ref. 16 reports that US NRC and Westinghouse performed separate modelling of the limiting sequence assuming ambient heat losses and both showed that the RCS remains subcooled for a period exceeding 72 hours. The only impact on the design basis accident transient results of modelling the ambient heat losses (and using RELAP rather than LOFTRAN) is a faster RCS cooldown rate due to increased heat removal.

90. I am satisfied that Westinghouse has fully addressed the concern raised by Amec Foster Wheeler about ambient heat losses (note, Ref. 15 was finalised before Westinghouse addressed the issue in updated documentation). I welcome both the use of RELAP to investigate the known limitations of LOFTRAN and the reference to experimental APEX test rig results. My confidence in Westinghouse's conclusions is also strengthened by the rigour of US NRC's evaluation detailed in Ref. 16.

91. Crucially, I am satisfied through the work identified in Ref. 18 that Westinghouse has adequately met the basic requirement of the GDA Issue to provide validation evidence that the IRWST is functionally capable of cooling the PRHR during intact circuit faults for 72 hours. I am also satisfied that it has met its own declared objectives for Category A functions as set out in Chapter 5 of its PCSR (Ref. 19):

> Category A safety functions are those utilised to achieve and maintain a non-hazardous, stable state for at least 72 hours following an initiating event. Category A safety functions are fulfilled by those systems analysed in the plant [design basis accident analysis]

92. For substantiating this aspect of the safety case, it is my judgement that appropriate conservatisms have been included in Westinghouse's analysis of the limiting event, consistent with its 'normal' shorter-term design basis analysis and the requirements of SAP FA.6. I am reassured that Amec Foster Wheeler (Ref. 15) and US NRC (Ref. 16) make similar conclusions. I will return to the subject of uncertainties in Subsection 4.6 below.

93. It is worth noting that the transient analyses of intact circuit design basis faults presented in Chapter 9 of the PCSR (Ref. 19) are still those generated by the short-term analysis methods as used in Chapter 15 of the EDCD (Ref. 2). For example, the analysis of the loss of normal feedwater fault with a coincident loss of AC power is not from the new work presented in Ref. 18. Instead, the demonstration of the ability to reach a safe, stable state is provided in Appendix 9C of Ref. 19. I am content with this approach.[7]

---

[7] Westinghouse has updated its methodologies from those presented in Ref. 2, providing a new generation of design basis analyses. The adequacy of this new analysis has been assessed outside this report as part of ONR's assessment of GI-AP1000-FS-02 (Ref. 20). One notable change from the Ref. 2 analysis is a conservative assumption for the containment pressure. This addresses a comment raised in Ref. 1 about the potential impact of pressure on IRWST behaviour, while avoiding the need for a coupled calculation similar to that used in LOCA analyses or the extended transients considered in this report.

4.5 TSC Observations on the Selection of the Limiting Transient

94. In Ref. 15, Amec Foster Wheeler agreed with Westinghouse's claim that for most intact circuit design basis faults the most challenging time with respect to margins to safety limits (eg the margin to DNBR limit) is in the period immediately following an initiating event and that the long-term status of the IRWST water level is irrelevant to the modelling of this period of the transient. Amec Foster Wheeler also recognised that Westinghouse had chosen a limiting transient (along with pre-fault containment conditions) to maximise the challenge for PRHR heat exchanger performance. However, it took the view that alternative assumptions could be more limiting with regard to demonstrating a margin to overfilling the pressuriser.

95. Amec Foster Wheeler gave two main reasons for why it believed this to be an important point:

- Compliance with the pressuriser level limit throughout the 72-hour period is an appropriate metric for demonstrating that the RCS is under the control of the frontline safety systems, such that an unsafe condition does not arise.
- Should the pressuriser limit be exceeded and the pressuriser safety relief valves open, the RCS pressure could drop sufficiently for the saturation point to be reached and the resulting two-phase flow could challenge the effectiveness in the PRHR heat exchanger.

96. I asked Westinghouse through a regulatory query to respond to these observations (Ref. 23). Westinghouse acknowledged the importance of the pressuriser overfill success criterion for frequent intact circuit faults. It stated that the objective of the criterion is to avoid the pressuriser safety relief valves opening and relieving water rather than steam, effectively escalating the event to a LOCA fault.[8]

97. Amec Foster Wheeler identified a loss of normal feedwater fault (with AC power available) as a more onerous fault for pressuriser overfilling than the limiting event selected by Westinghouse in Ref. 18. In the PCSR Chapter 9, Westinghouse actually identifies a further two events that are also challenging: inadvertent actuation of a CMT and a chemical and volume control system malfunction. However, for all three events, Westinghouse's design basis safety case already credits the operator opening the Class 1 reactor vessel head vent (30 minutes after the 'high-2' pressuriser water level set-point is reached) to ensure that water does not pass through the pressuriser safety relief valves.

98. In my view, Westinghouse's submissions for GI-AP1000-FS-06 would have been stronger if it had provided more information on the limiting faults for margin to pressuriser overfill and commented in the relevant sections of PCSR Chapter 9 (Ref. 19) that the timing of actions for the three limiting events could be altered if it had used its modified condensate return methodology. However, I have no objections with Westinghouse's safety case claim that operator actions (on extended timescales and prompted by numerous indications) can be used to ensure that the pressuriser does not go water solid (addressing Amec Foster Wheeler's first challenge) and I am content with Westinghouse's demonstrations of the PRHR heat exchanger's effectiveness even with two-phase flow (addressing Amec Foster Wheeler's second challenge). I also observe that it is highly unlikely that an operator would choose to manage an intact circuit fault for an extended period of time just using the PXS (long enough for PRHR heat exchanger uncovery to be of relevance) if AC power is

[8] Westinghouse comments in Ref. 23 that the SSCs to protect against LOCA faults remain available. Westinghouse also notes that in all extended transients just using the PRHR, eventually the RCS pressure and temperature will drop such that the steam bubble collapses and the pressuriser goes water solid. However, at this time in the transient the RCS pressure is much lower than the relief valve set-point.

available. Therefore, it seems reasonable to select a loss of normal feedwater fault without AC power for consideration, in preference to the similar fault with AC power available.

4.6 Definition and Demonstration of a Stable, Safe State

99. The expectation that the safety case will show that safety measures are capable of bringing a nuclear facility to a stable, safe state following any design basis fault is long established in the UK and set out in SAP FA.8 (Ref. 10). Similar requirements exist in other countries, including the US where the **AP1000** design was developed. However, early on in the development of the **AP1000** reactor (and its predecessor, the AP600 reactor design), it was recognised that the proposed passive systems would not be able to bring the plant to the cold shutdown conditions normally achieved on nuclear power plants crediting active cooling systems. Following extensive dialogue with the Electric Power Research Institute (EPRI), vendors (notably Westinghouse) and the ACRS, US NRC set out its policy position for the requirements on safe shutdown for passive plants in SECY-94-084, Item C (Ref. 33).

100. Ref. 33 resulted in several requirements for safe shutdown, which the EDCD (Ref. 2) sets out to demonstrate:

- A stable, safe shutdown condition was defined as an RCS temperature of 215.6°C (420°F).
- The passive safety systems were required to have sufficient capacity to reduce the RCS temperature to 215.6°C (420°F) in 36 hours.
- The passive safety systems were required to be able to maintain stable, safe conditions for at least 72 hours without makeup water following reactor shutdown.
- A long-term safe, stable condition needed to be maintained beyond 72 hours by simple, unambiguous operator actions, easily accomplished by lower safety class equipment (potentially brought in from off site).

101. ONR expressed no concerns or objections to these requirements in the GDA Step 4 assessment (Ref. 1), other than the need for Westinghouse to address GI-AP1000-FS-06. However, following the writing of GI-AP1000-FS-06 and Westinghouse's development of its new methodology for modelling condensate return fractions, it became more difficult and nuanced for Westinghouse to demonstrate all these objectives. However, it also became apparent that not all of these were 'regulatory requirements' in the US or linked to definitive safety limits. Notably, the temperature definition of 215.6°C (420°F) and reaching it in 36 hours was traced to an EPRI objective for new plants rather than a US NRC regulation. In the UK, there is no prescription for the definition of safe, stable state given in the SAPs. It is for the safety case owner to set out its claims and arguments, justified with appropriate evidence; in this case what constitutes a stable, safe state and how it has been achieved.

102. As discussed in Subsection 4.4, I am satisfied that Westinghouse has been able to demonstrate through conservative analysis of the limiting intact circuit fault that stable, safe conditions can be maintained for at least 72 hours. However, it has not been able to show through this conservative analysis that 215.6°C (420°F) will be reached within 36 hours and maintained indefinitely.

103. In Ref. 18 and its supporting references, Westinghouse has put a lot of effort into demonstrating that 215.6°C (420°F) can be reached in 36 hours. However, it has only been able to do this by relaxing the level of conservatism in its LOFTRAN model from that used in its 'normal' design basis analysis. Significantly, it assumes nominal values for initial reactor power and decay heat respectively. I welcome the insights into the expected plant behaviour provided by this analysis, and I also note that Ref. 16 states that US NRC has independently modelled this aspect of the **AP1000** plant and

obtained similar results. However, in my opinion, the need to demonstrate the identified temperature in the specified time is rather arbitrary and secondary to the principal safety case demonstration provided by the 72-hour analysis using conservative assumptions. Therefore, I do not have any concerns about the need to relax the assumptions in the analysis and I am comfortable with the reduced prominence of this claim in the updated UK **AP1000** safety case as set out in Appendix 9C of Ref. 19.

104. The EDCD (Ref. 2) stated in many places that the PXS could adequately cool the RCS indefinitely following an intact circuit fault, with just a small number of simple operator actions required after 72 hours to replenish the PCCWST water being poured on the outside of the containment. However, as stated in Section 3 above, Westinghouse has now modified this claim in the PCSR to be “at least 14 days”.

105. The analysis that supports this revised claim uses the same relaxed assumptions as those made in the 36-hour analysis but I have no objections to this. The analysis retains some conservatism and it demonstrates that there are no ‘cliff-edge’ phenomena that occur at 14 days. Instead, the analysis shows that it would take about 28 days for the PRHR heat exchanger to become uncovered. This significantly exceeds any expectations I have for considering delays in restoring power and active cooling systems. Crucially, Westinghouse has addressed the concerns about the PRHR heat exchanger being able to work effectively with two-phase flow (expected to occur after 72 hours but before 14 days) and LOFTRAN’s ability to model the relevant phenomena out to this length of transient (Ref. 24). In my opinion, 14 days appears to be a reasonable claim to make on the capability of the PRHR based on Westinghouse’s analysis but there is no UK expectation that establishes such a duration or a need for it to be demonstrated with a particular level of conservatism.

106. At any point during an extended intact circuit fault, the **AP1000** plant has ‘in reserve’ the ability to achieve a safe shutdown state using a passive feed-and-bleed approach should a problem occur with the PRHR / IRWST systems (for example, it is discovered that the RCS is losing inventory). The bleed, release of RCS energy, is performed by manual actuation of the Automatic Depressurisation System (ADS). The feed, RCS inventory makeup, is performed by the accumulators and IRWST gravity injection (and in the longer term by the recirculation of water flooding the lower levels of the containment). To maintain this capability, there are a number of actions the operators need to take to ensure that the Class 1 DC batteries remain available, firstly between 24 and 72 hours, and then after 72 hours. However, I am satisfied that these actions are clearly identified in the **AP1000** safety case, notably Appendix 9C of PCSR Chapter 9 (Ref. 19).

107. As stated in Section 3, when Westinghouse extended out its conservative analysis of the intact circuit transient beyond 72 hours (ie with design basis assumptions), it found it would take about seven days for the effectiveness of the PRHR heat exchanger to reduce sufficiently for the RCS temperature to start rising. Although Westinghouse’s revised position is that 215.6°C (420°F) is an arbitrary temperature not linked to a safety limit, it has recognised that there does need to be an upper temperature established which informs the operator’s judgement on whether the PRHR is adequately cooling the RCS. As an addition to the PCSR demonstration of the achievability of a safe, stable state (Ref. 19), it has defined 296°C (565°F) as such a limit to be included in the emergency procedures as a prompt for initiating passive feed-and-bleed. Again, this temperature is somewhat arbitrary; it has been chosen because it bounds the ‘no load’ temperature that is used in normal operations and therefore will be familiar to the operators.

108. I am content with this revision of the safety case. The adequacy of open loop cooling following LOCA faults was assessed during GDA Step 4 (Ref. 1). Triggering the ADS after 72 hours will be a similar but less onerous transient for the PXS systems to deal

with than a LOCA fault occurring at-power or shortly after shutdown. However, while I accept the safety case arguments put forward, I do think it is important that acknowledge that operator actions to monitor the RCS temperature and maintain an ability to initiate the ADS have an increased prominence in the safety case (although they were always there).

109. The first three stages of the ADS that would be utilised by the feed-and-bleed approach would result in RCS inventory being discharged into the IRWST. In Ref. 16, US NRC discusses a potential concern it identified with initiating the ADS if the IRWST water level had been allowed to drop below the level of the ADS spargers. The response it received from Westinghouse (Ref. 34) sets out multiple reasons why this is not a concern including:

- Vents above the IRWST are sufficient to release steam discharged from the ADS and prevent IRWST overpressurisation.
- The increase in containment pressure is bounded by the mass / energy releases considered during LOCAs or main steam line break faults which the **AP1000** containment has been designed for.
- ADS performance is improved if the spargers are uncovered.
- The pressure within the spargers will be significantly less than the normal RCS operating pressure if the ADS is actuated from a position where the PRHR heat exchanger is uncovered.

110. US NRC states that it was ultimately satisfied with the response it received and as a result it is content that there is no minimum IRWST water level required for ADS actuation that could potentially limit the duration of operation with the PRHR. I, too, am satisfied that ADS sparger uncovery does not impose a time constraint on the maintenance of a stable, safe state using the PRHR, but I have not discovered any appropriate discussion of this issue in UK-specific safety case documentation. I have needed to make use of the publicly available US correspondence on this matter (Refs 16 and 34) to come to a view on the adequacy of the position, but this slight shortfall in the completeness of the UK safety case is not sufficient to prevent the closure of this GDA Issue.

4.7 Adequacy of the PCSR

111. As part of its response to this GDA Issue, Westinghouse has updated Appendix 9C of the PCSR (Ref. 19). I have reviewed this section of the PCSR and found it to be an adequate summary of the safety case for the **AP1000** reactor's ability to achieve a stable, safe state following an intact circuit fault. It also provides appropriate links to the more detailed supporting analysis and test results that have been submitted to ONR for this GDA Issue but which have not been included in full in the PCSR.

112. I have also looked for evidence that Westinghouse has updated other chapters of the PCSR in response to the safety case changes resulting from this work, in particular that statements suggesting the PRHR can maintain the RCS below 215.6°C (420°F) for an indefinite period of time following an intact circuit fault. Westinghouse has demonstrated to my satisfaction it has made such alterations, with notable changes being made to Chapters 6 (Plant Description and Operation) and 20 (Structural Integrity).

113. DCP APP-GW-GEE-3692 (Ref. 25) introduces a new Class 1 downspout and piping network into the design. It was my starting expectation that a Class 1 SSC should feature in the 'Engineering Schedule' presented in Chapter 15 of the PCSR (Ref. 19). However, while the debris screens attached to the downspouts are listed, the downspouts themselves are not. I challenged Westinghouse on this apparent omission however it stated that has excluded all pipework (even if it is Class 1) from Engineering Schedule to prevent it becoming unmanageably long. The contents and adequacy of

the Engineering Schedule are beyond the scope of this fault studies assessment however I accept the logic of not including the downspouts if consistency is going to be maintained with the general approach of not listing pipework. Westinghouse does not use the PCSR (including the engineering schedule) to control the **AP1000** design. Instead it uses 'system specification documents' which in turn reference 'piping and instrumentation diagrams' to define the design. I am content that the relevant system specification document impacted by the change have been identified by Ref. 25, and as stated in Sub-section 4.2, I have made a positive determination that Ref. 25 and other related DCPs are all included in the design reference report (Ref. 27).

4.8 Assessment Findings

114. Assessment findings are matters that do not undermine the generic safety submission and are primarily concerned with the provision of site-specific safety case evidence, which will usually become available as the project progresses through the detailed design, construction and commissioning stages.

115. Residual matters are recorded as assessment findings if one or more of the following apply:

- site-specific information is required to resolve this matter
- the way to resolve this matter depends on licensee design choices
- the matter raised is related to operator-specific features / aspects / choices
- the resolution of this matter requires licensee choices on organisational matters
- to resolve this matter the plant needs to be at some stage of construction / commissioning

116. In my assessment, I did not find any examples of matters that meet these criteria.

5 CONCLUSIONS

117. This report presents the findings of the assessment of GDA Issue GI-AP1000-FS-06 relating to the **AP1000** GDA closure phase.

118. Westinghouse has undertaken a significant amount of work to address this GDA Issue, including physical testing, design changes, the development of a new analysis methodology, and revisions to its safety case for demonstrating that a safe, stable shutdown state can be achieved by the **AP1000** PXS for intact circuit faults.

119. Through a combination of:

- my own assessment of the supplied submissions against the expectations of the SAPs;
- multiple meetings and discussions with Westinghouse over many months;
- a detailed assessment of the analysis and proposed design changes undertaken by a TSC commissioned by ONR; and
- consideration of the detailed assessment of the same issue performed by US NRC,

I am satisfied that Westinghouse has addressed the requirements of this GDA Issue and therefore GI-AP1000-FS-06 can be closed. There are no residual matters to be recorded as assessment findings that will need to be addressed by a future licensee.

6 REFERENCES

1.	Step 4 Fault Studies – Design Basis Faults Assessment of the Westinghouse **AP1000** Reactor, ONR-GDA-AR-11-004a Revision 0, November 2011, TRIM Ref. 2010/581406
2.	**AP1000** European Design Control Document, EPS-GW-GL-700 Revision 1, March 2011, TRIM Ref. 2011/81804
3.	GDA Issue "Validation of the IRWST Cooling Function for the PRHR", GI-AP1000-FS-06, www.onr.org.uk/new-reactors/reports/step-four/westinghouse-gda-issues/gi-ap1000-fs-06.pdf
4.	UK **AP1000** Assessment Plan for Closure of GDA Fault Studies Issues 1 to 8, ONR-GDA-AP-14-002 Revision 0, March 2015, TRIM Ref. 2015/51535
5.	ONR Guidance on Mechanics of Assessment, TRIM Ref. 2013/204124
6.	GDA Guidance to Requesting Parties, www.onr.org.uk/new-reactors/ngn03.pdf
7.	The Purpose, Scope, and Content of Safety Cases, NS-TAST-GD-051 Revision 4, www.onr.org.uk/operational/tech_asst_guides/ns-tast-gd-051.pdf
8.	GDA Issue "PCSR to Support GDA", GI-AP1000-CC-02 Revision 3, www.onr.org.uk/new-reactors/reports/step-four/westinghouse-gda-issues/gi-ap1000-cc-02.pdf
9.	Purpose and Scope of Permissioning, NS-PER-GD-014 Revision 5, TRIM Ref. 2015/304735
10.	Safety Assessment Principles for Nuclear Facilities, 2014 Edition Revision 0, ONR, November 2014, www.onr.org.uk/saps/saps2014.pdf
11.	IAEA Standards and Guidance: International Atomic Energy Agency (IAEA) Safety Standards Series – Safety of Nuclear Power Plants: Design, Specific Safety Requirements (SSR) 2/1 Revision 1, 2016 International Atomic Energy Agency (IAEA) Safety Standards Series – General Safety Requirements (GSR) Part 4: Safety Assessment for Facilities and Activities, 2007 International Atomic Energy Agency (IAEA) Safety Standards Series – Safety Guide: Safety Assessment and Verification for Nuclear Power Plants 2001 (this publication has been superseded by GSR Part 4 and SSG-2) www.iaea.org
12.	Western European Nuclear Regulators Association: Reactor Safety Reference Levels for Existing Reactors, September 2014 Statement on Safety Objectives for New Nuclear Power Plants, November 2010 Statement and Report on Safety of New NPP Designs, March 2013 www.wenra.org
13.	Specific Safety Guide No. SSG-2: Deterministic Safety Analysis for Nuclear Power Plants, IAEA, 2010, www-pub.iaea.org//MTCD//publications//PDF//Pub1428_web.pdf

14.	US NRC Regulatory Guide 1.203, Transient and Accident Analysis Methods, December 2005, www.nrc.gov/docs/ML0535/ML053500170.pdf
15.	Review of Westinghouse's Validation Evidence for the Effectiveness of the **AP1000** Passive Reactor Heat Removal System, 203171-TR-000068 Issue 01, February 2016, TRIM Ref. 2016/47823
16.	US NRC Advanced Safety Evaluation Report for Chapter 21 of the Levy Nuclear Plant Units 1 and 2 Combined License Application: "Design Changes Proposed In Accordance With ISG-11", March 2016, www.nrc.gov/docs/ML1608/ML16081A127.pdf
17.	US NRC Advisory Committee on Reactor Safeguards: Exemptions to the AP1000 Certified Design Included in the Levy Nuclear Plant Units 1 and 2 Combined License Application, April 2016, www.nrc.gov/docs/ML1610/ML16102A149.pdf
18.	Condensate Return Analysis Summary Report, UKP-PXS-GLR-001 Revision 0, August 2016, TRIM Ref. 2016/333124
19.	**AP1000** Pre-Construction Safety Report, UKP-GW-GL-793 Revision 1, January 2017, TRIM Ref. 2017/43700
20.	GDA Issue "Design Reference Point and Adequacy of Design Basis Analysis", GI-AP1000-FS-02, www.onr.org.uk/new-reactors/reports/step-four/westinghouse-gda-issues/gi-ap1000-fs-02.pdf
21.	**AP1000** Plant Safe Shutdown Duration Evaluation, APP-SSAR-GSC-009 Revision 0, June 2015, TRIM Ref. 2015/249293
22.	**AP1000** Condensate Return Test Identification and Phenomena, APP-PXS-GER-003 Revision 0, June 2013, TRIM Ref. 2015/249231
23.	Westinghouse Feedback on Amec Foster Wheeler Assessment of GI-AP1000-FS-06 Submissions, RQ-AP1000-1565, TRIM Ref. 2016/206289
24.	**AP1000** Long-term Passive Residual Heat Removal (PRHR) Operation Event Phenomena Identification and Ranking Table (PIRT), APP-SSAR-GSR-002 Revision 0, TRIM Ref. 2015/249297
25.	Changes for Condensate Return to IRWST, APP-GW-GEE-3692 Revision 0, December 2012, TRIM Ref. 2015/346133
26.	Polar Crane Girder Modification as a Result of Containment Condensation Return, APP-GW-GEE4657 Revision 0, December 2013, TRIM Ref. 2015/346275
27.	AP1000 Design Reference Point for UK GDA, UKP-GW-GL-060 Revision 10, TRIM Ref. 2017/18158
28.	TSC Assessment of GI-AP1000-FS-06 Submissions: Sensitivity of DBA to Assumed Value of Condensate Return Rate, RQ-AP1000-1395, TRIM Ref. 2015/364324
29.	PRHR Heat Exchanger Tube Plugging and PXS Condensate Return Analysis Changes, APP-GW-GEE-4651 Revision 0, December 2013, TRIM Ref. 2015/346271
30.	Update to Condensate Return Analysis, APP-GE-GEE-5007 Revision 0, April 2015, TRIM Ref. 2016/7964
31.	PXS Condensate Return: Identification of a Limiting Scenario, APP-PXS-M3C-079, Revision 0, TRIM Ref. 2016/380370
32.	**AP1000** Safe Shutdown Temperature Evaluation, APP-SSAR-GSC-536 Revision 3, April 2015, TRIM Ref. 2015/249282

33.	US NRC: Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs, SECY-94-084, March 1994, https://www.nrc.gov/docs/ML0037/ML003708068.pdf
34.	Duke Energy Letter to US NRC: Levy Nuclear Plant, Units 1 and 2, Docket nos. 52-029 and 52-30, Supplement 2 Response to NRC RAI letter 116 – SRP Sections 6.3 and 15.2.6, 15 January 2016, https://www.nrc.gov/docs/ML1602/ML16021A188.pdf